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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL. (Perry Nuclear Power Plant, Units 1 and 2)	))))))))	Docket Nos. 50-440 OL 50-441 OL

NRC STAFF PARTIAL ANSWER TO SECOND SET OF INTERROGATORIES TO NRC STAFF BY THE SUNFLOWER ALLIANCE

The following are the NRC Staff's voluntary answers in partial response to the second set of interrogatories by the Intervenor Sunflower Alliance (Sunflower).[1/]

Although Sunflower's filing has not attempted to make a showing of how these interrogatories to the Staff satisfy the requirements of 10 C.F.R. § 2.720(h)(2)(ii), the Staff has in this instance voluntarily answered certain interrogatories, and is providing documents, in the interest of assuring the development of an adequate record.[2/] The response to some interrogatories has been deferred

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1/ Sunflower Alliance's Interrogatories to the Nuclear Regulatory Commission Staff (Second Set), dated April 30, 1982 ("Interrogatories").

2/ Of course, the Staff is not waiving its right to object to discovery, e.g., pursuant to § 2.720(h)(2)(ii), now or in the future, by this voluntary response.

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until the Staff has completed its review. For certain other interrogatories to which the Staff is not filing a response, reasons are stated why the interrogatory would be objectionable even if properly filed before the Licensing Board pursuant to § 2.720(h).

In compliance with the Licensing Board's Special Prehearing Conference Memorandum and Order LBP-81-24, 14 NRC 175, 230-31 (July 28, 1981), Counsel for NRC Staff has conferred by telephone with Sunflower's counsel on several occasions regarding Staff's answers to Sunflower's interrogatories.

Affidavits supporting the Staff's responses will be filed with a supplement to these responses.

Statement of Purpose: The following interrogatories are designed to assess the potential dangers of an unmitigated ATWS, the status of development and implementation of ATWS mitigation systems, and the degree of industry compliance with any regulations or other criteria concerning ATWS.

Interrogatory 1

Table 1, p. 11, of NUREG-0460, Vol. 4 lists various alternatives for ATWS plant modifications. Alternative 4A is to be implemented by all plants (other than early operating plants) by January 1, 1984 (p. 13, Vol. 4). However, NUREG-0460, Vol. 4, p. 54 contains the following statement: "Each plant for which conformance to Alternative 4A is deemed not practical because of constraints imposed by basic plant layout, diesel capacity, or completed seismically qualified structures, shall submit by December 31, 1980, the optimization study set forth in Section 2.4.1, including alternatives for achieving a level of safety equivalent to Alternative 4A. This alternative (sometimes called "Alternative 3 1/2" is intended for operating plants and those well along in construction. Duplicate plants at the same site may be modified identically, even if the second unit is not as far along in construction as to fall within this provision if the first unit qualifies."

(a) Explain the apparent contradiction between the statement on p. 13 and that on p. 54. Will or will not all plants (except early ones) be required to implement Alternative 4A by January 1, 1984?

(b) Define the phrase "level of safety equivalent to Alternative 4A." How is this degree of safety quantified or otherwise determined? What types of alternatives are there for achieving this level or safety?

(c) Define the phrase "well along in construction," either by percent completion of the plant as a whole or by the completion of specific systems or structures within the plant. Specifically, how would this phrase be defined for the Perry Nuclear Power Plant?

(d) Is the Perry Nuclear Power Plant, Unit 1, far enough along in construction, as defined above, to qualify for the consideration of "Alternative 3 1/2"? If so, has the Applicant submitted the optimization study required? If so, produce this optimization study.

(e) If the Perry Nuclear Plant, Units 1 or 2, was not far enough along in construction to qualify for Alternative 3 1/2, is it the Staff's opinion that PNPP will be required to implement Alternative 4A in its entirety? When will this hardware be required to be installed? How is the Applicant's compliance to be insured?

(f) If PNPP Unit 1 is far enough along in construction to qualify for Alternative 3 1/2, will Unit 2 be permitted to be modified identically? Why has this provision (identical modification of duplicate same-site units) been included? Discuss why this provision will not degrade plant safety in the more easily modified second unit.

Response

Interrogatory 2

One of the modifications included in Alternative 4A for GE BWRs is modification of the scram discharge volume. Describe this modification in detail for the Perry Plant BWR/6 model. Has this been done at Perry?

Response

The Perry Plant BWR/6 model does not require modification to the scram discharge volume for Alternative 4A. The Perry design incorporates two separate scram discharge volume headers with an integral instrumented volume at the end of each header, thus providing close hydraulic coupling. Additionally, each instrumented volume has redundant and diverse level instrumentation for the scram function attached directly to the instrumented volume. The Staff's evaluation of the Perry design concerning the scram discharge volume is provided in the Safety Evaluation Report dated May 1982 (Section 4.6).

Interrogatory 3

Where does the PNPP design presently stand in regard to the alternatives listed in NUREG-0460, Vol. 4? E.g., does the current plant design implement Alternative 2A, 3A, or 4A?

Response

Interrogatory 4

What constitutes scram failure in a BWR/6 such as Perry? E.g., describe the combination of the following failures which will result in the loss of control of reactivity and failure to attain hot shutdown: insufficient rod insertion speed, percent of length withdrawn which results in failure, number and location of failed rods which results in scram failure.

Response

Interrogatory 5

Describe in detail, along with their frequency of occurrence for each year of plant operation, any and all transients capable of initiating reactor scram in a BWR/6.

Response

Interrogatory 6

Describe all scram system failures, including common-mode failures, capable of producing ATWS in a BWR/6.

Response

Interrogatory 7

For each of the transients listed in #5 above (and for any transient not listed in the response to #5 but included in Table A.2, Vol. 4 of NUREG-0460), perform a time-domain analysis, specific to the Perry Plant, assuming that control rod scram does not occur, but that the recirculation pump trip does function and the SLCS, as presently designed, is manually operated. Assume all plant systems to be as currently described in the FSAR. Include in the analysis any and all plant systems and functions affected by ATWS and any consequences thereof, including but not limited to core integrity, containment integrity, suppression pool effects, reactor internals, ECCS functions, dilution of SLCS boron by ECCS, power oscillations, and offsite radiation doses to the public. Present the analysis in this manner: the transient begins at t=0; list time of occurrence for each major action or consequence during the ATWS (e.g., RPT, SLCS activation, containment isolation, and maximum and minimum values of the following parameters to be presented graphically) until such time as either the reactor is brought into cold shutdown or core melting occurs. List all assumptions made for operator actions. Present the following parameters graphically as a function of the time (use appropriate units and scales): neutron flux, power levels, RPV pressure, suppression pool temperature, containment pressure, steamline pressure, water level in RPV, heat flux, and fuel cladding temperature and radiation doses to public at site boundary, 5 mile radius, 10 mile, 20 mile, and 50 mile radius. Also perform the analysis as described above for the following conditions:

1) As above, only with automatic SLCS.

2) Full implementation of Alternative 4A.

Response

Interrogatory 8

How many transients occurred in each of the years 1978, 1979 and 1980?

Response

Interrogatory 9

Does PNPP have the recirculation pump trip initiated by high pressure? What other conditions can initiate the RPT? Explain how this feature mitigates the consequences of ATWS; about what % negative reactivity does the RPT contribute? When was (or will be) the RPT feature installed?

Response

Interrogatory 10

Show the RPT hardware conforms with the appropriate standard, below:

1) If installed before July 1, 1981, the approved Hatch or Monticello designs. (Supply the appropriate design)

2) If installed after July 1, 1981, Appendix C of Vol. 3, NUREG-0460.

Response

The RPT (Recirculation Pump Trip) circuitry at the Perry Nuclear Power Plant Units 1 and 2 was designed prior to July 1, 1981 and is currently being installed. This circuitry however, is not required to conform to the proposed criteria in Appendix C (ATWS Equipment Requirments) of NUREG-0640, volume 3. The staff's current requirements for BWR ATWS RPT designs are found in Appendix A to a generic letter dated January 8, 1979 which required all operating BWRs to install a RPT prior to November 1, 1979. The Perry RPT design complies with these requirements. As stated in Section 7.2.2.3 of NUREG-0887 (Safety Evaluation Report related to operation of Perry Nuclear Power Plant, Units 1 and 2(), the RPT design consists nonsafety-grade circuitry that trips the recirculation pumps without transfer to the low-frequency motor-generator set upon receipt of a high reactor pressure or low reactor vessel water level (trip level 2) signal. This design is similar to the Grand Gulf and Hatch designs that the staff has reviewed and accepted.

If should be noted that the present staff position regarding RPT designs represent interim requirements pending the outcome of the proposed ATWS rule (SECY 80-409).

Interrogatory 11

Demonstrate that the ARI system meets the critiera of IEEE Standard 279, and that the RPT and SLCS logic meet the criteria of Appendix C of Vol. 3, NUREG-0460.

Response

The Staff has no requirements to date for installation of an ARI (ATWS Rod Injection) system (i.e., a system which is independent and diverse from the RPS and acts as backup to the electrical portion of the current scram system) or automatic actuation logic for the SLCS. Both an ARI system and SLCS logic modifications were proposed in Volumes 3 and 4 of NUREG-0460 (Anticipated Transients Without Scram for Light Water Reactors) and the ARI system would be required by the proposed ATWS rule (SECY 80-409), however, no requirements for these systems have been imposed pending outcome of the ATWS rule. The only ATWS modification required on both operating and OL BWRs to date is the RPT. The RPT design installed at Perry complies with current staff requirements as stated in the response to Question 10 above.

Interrogatory 12

Have the code verification tests for BWRs described on p. B-3 of Vol. 4, NUREG-0460 been performed? If not, why not, and when will they be performed? If so, what were the results of these tests?

Response

Interrogatory 13

Describe the effects of power oscillations, such as are described on P. A-67, Vol. 4, NUREG-0460, on fuel and containment integrity and any other affected system at PNPP.

Response

Interrogatory 14

Give a cost estimate for the installation of an automated standby liquid control system at PNPP, Units 1 and 2: provide documentation to support this estimate. Include in the estimate any necessary modifications to other systems, e.g., addition of sufficient diesel generator capacity. Also give a cost estimate for the complete implementation of Alternative 4A as described in Vol. 4 of NUREG-0460.

Response

Interrogatory 15

Give an estimate of the cost and downtime associated with the cleanup of a inadvertent activition of the SLCS. Give the waste storage tank capacity and evaporator capacity; compare these withother BWRs, and indicate how these systems are involved in the boron cleanup.

Response

The function of the Standby Control System (SLCS) is to shut down the reactor in the unlikely event the normal control system even becomes inoperative. It accomplishes this by pumping a boron absorber solution into the reactor vessel by means of manually-initiated high pressure pumps (See FSAR Section 9.3.5).

Assuming that activation of the SLCS has occurred, the water in the reactor vessel must be cleaned or decontaminated to remove the boron before the reactor can be operated. The SLCS is designed to achieve a maximum average concentration of 660 ppm of natural boron in the reactor core. This must be reduced to less than 15 ppm before the reactor can be restarted.

Removal of boron from the reactor coolant in a plant having the diversity of liquid treatment systems such as the Perry design, has several options in the treatment of boron-contaminated reactor coolant. The most probable approach would involve the operator flushing the reactor vessel with clean water from a reservoir such as the condensate storage tank. Once a large fraction of the boron has been removed, the remaining boron would be removed by recirculating the coolant through the Reactor Water Cleanup System (this procedure is desscribed in greater detail below).

The reactor vessel would first be flushed with clean water from the condensate storage system. The overflow is directed to the liquid radwaste system for processing and re-use. The liquid radwaste system has three subsystems capable of handling imputs from the reactor coolant system; these are the Equipment Drain Processing System, with two 36,500 gallon capacity collection tanks, the Floor Drain Processing System, with two 36,500 gallon capacity callection tanks, and the Chemical Waste Processing System, with two 19,00 gallon collection tanks.

With a nominal reactor vessel water volume inventory of 65,000 gallons, the collection tankage of the Equipment Drain and Floor Drain Processing Systems would accommodate two full-volume flushes of the reactor vessel, which would reduce the boron concentration by approximately a factor of four. The remaining boron in the reactor coolant cold then be removed by recirculation through the Reactor Water Cleanup System (RWCS), which is a normal-operation subsystem of the reactor's coolant water system. The RWCS at Perry has two filter-demineralizers using powdered mixed-bed ion in exchange resins as the cleanup medium.

The length of time required to remove enough boron to permit the reactor to be re-started is estimated to take approximately one week.

The FSAR is not specifc as to the path of the water initially flushed from the reactor at the start of cleanup. Based upon experience with other plants, however, we have assumed that the water is diverted to the collection tanks of the Liquid Radwaste Processing System. These tanks have a total capacity of 146,000 gallons, which is more than adequate to accommodate two complete volume changes of the reactor vessel water. Since this water cannot be reused in the plant until the boron is removed, the water must be treated to remove the boron. In our evaluation, we assumed that the water will be treated through one of two available evaporators at a rate of 30 gpm and further decontaminated or "polished" by passing the condensate through a mixed-bed in exchange demineralizer. While we cannot accurately predict Perry's costs of operation, we have assumed a cost of 15¢ per gallon for this form of treatment. After treatment, the water is available for re-use in the plant.

The principal cost to the plant operator is the loss of revenue dire to an unscheduled plant shutdown lasting seven days. The costs of decontaminating the reactor water are relatively small in comparison. At Perry-1, two 30 gpm evaporators are available in the liquid radwaste system. Assuming a $250 per hour per evaporator operating cost, as derived from Regulatory Guide 110, the approximate cost of evaporation is estimated to be about 15¢ per gallon, or a total of about $18,000 for 120,000 gallons of water. Since the RWCS is normally in operation, regardless of SLCS operation, there is no significant added cost associated with operation of this system, with the possible exception of a small number of extra resin changes, at a cost estimated to be about $200 per resin change.

A plant having the diversity of liquid treatment systems that Perry-1's design incorporates has several options other than the systems described above in the treatment of boron-contaminated coolant. While we have described evaporative treatment of the flush water, it would be possible to completely deborate the coolant using only the RWCS, to utilize the condemsate cleanup demineralizers, or to use only the demineralizers of the liquid radwaste system. The choice becomes one of operational convenience, time and cost of equipment operation and would likely be an operational decision based on plant conditions existing at the time of such an incident.

The Perry plant waste collector tank capacity of Perry-1 is similar to that of other BWRs of the same size. The corresponding systems at the Clinton and Grand Gulf plants have a capacity of 110,000 gallons or slightly less than that of Perry-1 Grand Gulf has two evaporators of 25 gpm capacity, while Clinton has three evaporators of 30 gpm capacity. As indicated in subpart 7 of our response to this interrogatory, Perry has two 30 gpm evaporators. Only one is normally needed, but a 60 gpm capability is possible using both evaporators.

<u>Interrogatory 16</u>

The FSAR describes the neutron poison used in the SLCS as sodium pentaborate, produced by dissolving stoichiometric quantities of borax and boric acid in demineralized water. These boron compounds are described as consisting of natural boron. Does this mean that the boron is a mixture of the 2 predominant isotopes in the ratio that they occur naturally, i.e., $5B^{10}$ at 19,78% and $5B^{11}$ at 80.22%? Boron-10 is the only isotope that is effective at neutron capture; thus, if the sodium pentaborate is 80% boron-11, is not the poison then only 20% effective? Has this effect been considered in the assessment of the SLCS capabilities?

Response

The intervenor understanding of the meaning of natural boron in the SLCS context is correct. Also, it is true that, for each atom of boron present, natural boron is only about one-fifth as effective as fully enriched boron-10. However, this fact has been taken into consideration in the design and analysis of the SLCS by adjusting the boron concentration in the water to obtain the correct abses from cross section. Since the boron concentration is far below the solubility limit (approximately 600 parts per million versus 10,000 parts per million) this poses no problems.

Interrogatory 17

Has the use of different chemicals as neutron poisons, e.g., gadolinium, been investigated? if not, why not? if so, what results?

Response

So far as the Staff is aware no consideration has been given to use of other salts containing high absorbtion isotopes in the SLCS. As indicated above, boron is effective, has wide application in the nuclear industry and has well understood behavior.

Interrogatory 18

Perform a value/impact analysis, like that in NUREG-0460, specific to PNPP for: (1) the automation of the SLCS, (2) complete implementation of Alternative 4A. Both modifications are assumed to be made during construction on both Units 1 and 2.

Response

Statement of Purpose: The following interrogatories on the ECCS issue are designed to discover more information and evidence that can be used as evidence on the hearing of this cause.

Interrogatory 19

Section 6.3.1.1.2 of the PNPP FSAR states that, as a minimum, the following equipment shall make up the ECCS:

1 High Pressure Core Spray
1 Low Pressure Core Spray
3 Low Pressure Coolant Injection Loops
1 Automatic Depressurization System

Does the Emergency Core Cooling System at Perry have any other systems above and beyond these minimum requirements?

Response

Interrogatory 20

The applicant's FSAR states (Sections 1.5.1.1, 3.9.2.4) that Perry is the prototype 238 size BWR/6 plant. Describe in detail any special, more stringent testing requirements for prototype plants, especially those pertaining to the ECCS.

Response

Interrogatory 21

In the opinion of the staff, has the ECCS evaluation model for General Electric 238 size BWR/6 met all the criteria of 10 CFR 50.46 and Appendix K of Part 50? If not, specify what parts of the evaluation model do not comply.

Response

Interrogatory 22

Which of the four ECCS component systems listed in [#19] would be under test in a full scale 30° sector steam test?

Response

The high pressure core spray system and the low pressure core spray system would be the systems tested in a full scale 30° sector steam test. These two spray systems are simulated in the 30° sector steam facility for the BWR/6 test to quantify the steam effect on the core spray distribution.

Interrogatory 23

Would the full scale 30° sector steam test involve other, auxiliary systems not directly related to the phenomena at question, but nonetheless necessary for ECCS operation under real conditions (e.g., diesel generators)?

Response

No. See response to Interrogatory number 24.

Interrogatory 24

In the opinion of the Staff, is full scale testing mandated by 10 C.F.R. Part 50. Appendix K?

Response

No. The goal of full scale 30° sector steam core spray test is to quantify the steam effect on the core spray distribution. It is a confirmatory test and is not mandated by 10 C.F.R. Part 50, Appendix K.

Interrogatory 25

Describe in detail any controversy concerning the independence and separability of thermal and hydraulic effects in the specific calculation used to demonstrate compliance of the GE 238 size BWR/6 (generic) and PNPP (specific) ECCS evaluation models with 10 C.F.R. Part 50. Appendix K.

Response

Prior to ASEA tests, GE had conducted full scale spray distribution tests in air at atmospheric pressure for all BWR's to ensure that the necessary minimum flow would be provided to each fuel bundle. The results of the ASEA tests raised a concern regard steam effect on the core spray distribution. Consequently, GE presented "interim" test results and calculations performed to justify acceptability of presently assumed core spray cooling during the "interim" period before completion of Task Action Plan (TAP) A-16. Of more fundamental concern is validity of the basic assumption inherent in the "interim" tests, which is the separability of hydrodynamic phenomena (droplet-to-droplet) interaction where spray patterns from two or more nozzles intersect) and thermal phenomena (steam condensation). This separability assumption is implicit in the "interim" results since only single nozzle tests have been performed in steam, and nozzle-nozzle interactions have been measured only in air. Single nozzle test results in steam support the separability assumption by indicating that most steam condensation occurs in the first six inches of spray flow outside the nozzle. Individual nozzles on a BWR core spray sparger are sufficiently separated so that their spray patterns do not intersect within the first six inches outside the nozzle. Therefore, the hydrodynamic and thermal effects should occur in separate regions, thereby supporting the basic assumption made by the "interim" tests. Nevertheless, multi-nozzle tests in steam have not been done in the "interim" test. In the TAP A-16, the multi-nozzle tests in steam have been done. The test results vigorously demonstrate the separability of the hydrodynamic and thermal effects on the core spray distribution. Based on the test results which are documented in NEDO-24712, we conclude that the core spray transfer coefficient used in the GE ECCS Evaluation Model has been demonstrated to be consistent with the requirement of 10 C.F.R. Part 50, Appendix K. (see response to question 27).

Interrogatory 26

Describe the status of Task Action Plan A-16: Produce any and all documents relating to TAP A-16 (see NUREG-0410).

Response

All the core spray tests under Task Action Plan A-16 have been completed. The results of core spray tests for a simulated BWR/6 core are documented in NEDO-24712 (August 1979) - Core Spray Design Methodology Confirmation Test. The results of the Staff's review on this test report is included in a letter from R.L. Tedesco (NRC) to G. Sherwood (GE), Acceptance for Referencing Topical Report NEDO-24712 - Core Spray Design Methodology Confirmation Tests. 1/30/81. Copies of the above documents will be made available at a later date.

Interrogatory 27

What bearing has TAP A-16 upon the PNPP ECCS evaluation?

Response

In its evaluation of NEDO-20566 Amendment 3, "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 C.F.R. 50 Appendix K - Effect of Steam Environment on Core Spray Distribution," the Staff concluded that little degradation would occur in heat transfer for spray flow as low as 0.38 times the flow necessary to remove decay heat by vaporization, or approximately one gallon per minute. (APED-5529 - Core Spray and Core Flooding Heat Transfer Effectiveness in Full-Scale Boiling Water Reactor Bundle, June 1968, F. Schrank and J. Leonard). The heat transfer coefficients in the GE Evaluation Model are based on BWR FLECHT data and a minimum bundle flow of one gpm would justify the heat transfer coefficient for core spray cooling of 1.5 Btu/hr-fi^{2}-° F, which is consistent with the 10 C.F.R. 50, Appendix K requirement and used in that model. As indicated in NEDO-24712, (see response to Interrogatory 26, above) the minimum bundle flow for a BWR/6 is 2 gpm. Based on this information, the Staff found that the GE ECCS Evaluation model demonstrates compliance with the 10 C.F.R. 50 Appendix K requirement for core spray cooling for a BWR/6. The Staff, therefore, approved the application of the GE ECCS Evaluation for the analysis of the PNPP ECCS.

Interrogatory 28

Describe in detail the ECCS tests in Europe that demonstrated unpredicted effects and which led, in part, to the formulation of TAP A-16. Explain how these tests related to PNPP.

Response

During 1974, ASEA-ATOM, a Europe manufacturer of BWR's, indicated that the nozzle spray patterns were effected by several test parameters, including the system pressure and nozzle flow rate. More specifically, the characteristic core angles of spray nozzles tested in the ASEA facility narrowed in a steam environment by comparison to the core angles observed in air. (The detail of these tests is described in the following report: Section 3 of NEDO-20566-3, Effect of Steam Environment on BWP Spray Distribution, dated April 1977.

Based on the results of spray nozzles test conducted in the ASEA facility TAP A-16 was established to quantify the effects including the steam effect on the spray distribution. The results of tests of core spray distribution from BWR/6 nozzles are documented in NEDO-24712, August 1979. The Staff would expect that the results of these core spray tests would be applicable to PNPP.

Interrogatory 29

In the opinion of the Staff, to what degree is core spray distribution predictable?

Response

As stated in the safety evaluation report in a letter from R.L. Tedesco (NRC) to G.Sherwood (GE), Acceptance for Referencing Topical Report NEDO-24712-Core Spray Design Methodology Confirmation Tests dated January 1, 1981, the Staff's position is as follows:

> Although error bands on the pre-test predictions are quite large, actual data compare very well with predicted point. Close scrutiny of the data shows pressure and steam flow effects to be small but not negligible. Interaction effects obtained from air tests with simulator nozzles do predict the shift in location

of minimum bundle flow from single sparger tests to double sparger tests and actually predict the correct radius for minimum bundle flow. Therefore, the SSTF tests constitute a confirmation of the GE spray distribution design methodology for BWR/6 type spargers.

Interrogatory 30

In the opinion of the staff, is the ADS entirely sufficient and functional in all its expected operational modes?

Response

Interrogatory 31

Name any BWRs in this country which have been required to perform a full scale 30 degree sector steam test before being authorized for operation by the NRC.

Response

None.

Statement of Purpose: The following interrogatories on the Emergency Planning issue relate the effectiveness and feasibility of Applicant's Emergency Planning and are designed to discover evidence to be used a the hearing.

Interrogatory 32

Produce any and all documents, correspondence, or memoranda between CEI and the NRC, FEMA, local or state governments or any other entity relating to the use of thyroid blocking agents by CEI employees, emergency workers, or the general public.

Response

Interrogatory 33

Demonstrate and discuss how emergency response facilities meet each and every criterion listed in NUREG-0814; answer all questions therein. (Emergency response facilities include the control room, Technical Support Center, Operational Support Center and Emergency Operations

Facility).

Response

Interrogatory 34

NCRP Report No. 55 at pp. 16-17 indicates that engineered safeguards at reactors may reduce the release of radioiodine during a nuclear accident. For each safeguard listed therein (and below), describe the system, if any, that will be in place at PNPP, explain how the system works to reduce iodine release, and indicate how efficient said system is at reducing radioiodine levels.

(1) various methods for condensing the radioiodine-bearing steam that would be released to the reactor building.

(2) enclosing the reactor in a sealed containment structure.

(3) recirculating the contained atmosphere through absorbents and filters that remove radioiodines.

(4) operation of sprays containing chemicals capable of absorbing the radioiodines and reducing their concentration in the atmosphere of the containment building.

Response

Interrogatory 35

Explain how the plume exposure pathway EPZ depicted in Figure II-2 of Appendix D of Appendix 13A of FSAR was derived. Explain precisely how each and every one of the following factors was considered in the determination of the extent of the plume exposure EPZ; demography, including permanent and seasonal residents and transients; meteorology; topography; land use characteristics; access routes; local jurisdictional boundaries; release time and energy characteristics; release height; radionuclide content of release, including release fractions; plume dispersion, including plume rise; deposition velocity; dose-effects; sheltering and shielding; radiation treatment; breathing rates; time of year of release.

Response

Interrogatory 36

Does Staff agree with this EPZ area?

Response

The Staff has not yet formulated its position on the adequacy of the plume exposure pathway EPZ, and is awaiting the FEMA evaluation on this matter.

Interrogatory 37

Describe in detail the methods and standards by which the evacuation time estimates contained in Tables V-4 and V-5 in Appendix D of Appendix 13A of FSAR were evaluated.

Response

The answer to this question will be provided later. Background information on this subject can be found in NUREG/CR-1745 (BHARC-401/80-017), "Analaysis of Techniques for Estimating Evacuation Times for Emergency Planning Zones."

Interrogatory 38

What consequences (including early fatalities, delayed fatalities, early injuries, delayed injuries, developments and genetic birth defects, and land and water contamination) would be associated with each of the evacuation time estimates listed in the FSAR?

Response

The probabilities and consequences associated with an accident at the Perry plant, in which an overall evacuation time estimate for the 0-10 mile EPZ is used, are presented in Section 5.9.4 of the Perry DES (NUREG-0884). The evacuation model used is described in Appendix F to the DES. The Staff's evacuation time estimates are partly based upon preliminary evacuation time estimates generated by the Applicant, but the

Staff does not perform a separate consequence analysis for each of the evacuation time estimates listed in the FSAR.

Interrogatory 39

What range of estimates would be acceptable to the Staff? State specifically the highest evacuation time estimage which would be acceptable to the Staff for each evacuation sub-area drawn in the FSAR, and describe the consequences (as categorized above) which would be associated with each such evacuation time.

Response

The Staff does not determine the acceptability of evacuation times, per se, but does evaluate the acceptability of the methodology used by applicants in determining evacuation times. Therefore, the Staff does not have a highest evacuation time estimate which is acceptable to it.

Interrogatory 40

Has the Staff or Applicant (or anyone on their behalf or to their knowledge) conducted any generic or site-specific consequence analysis for (or having relevance to) releases from PNPP equivalent to the BWR-1 to BWR-4 releases defined in WASH-1400? If so, set forth in detail the methodology, assumptions, and results of any such study, including calculations of early fatalities, delayed fatalities, early injuries, delayed injuries, developmental or genetic birth defects, and land and water contamination. If not, by whom was the decision made that such a study was unnecessary and what were the reasons for that decision? What process was followed in reaching that decision?

Response

Interrogatory 41

Has the Staff or Applicant (or anyone on their behalf or to their knowledge) conducted any generic or site-specific accident consequence analysis for accidents with containment failure modes such that the radioactive releases exceed those set forth in the design basis accident assessment described in Chapter 15 of the PNPP FSAR? If so, answer Interrogatory #40, specific to any such study.

Response

Interrogatory 42

In the Staff's opinion, is it possible to evacuate safely the total permanent, seasonal, and transient populations within each of the following areas at any time of day or any time of year? Describe in detail any assumptions made and indicate how your response would differ if that assumption were changed. Disclose any assumptions made with respect to an acceptable level of risk to the evacuating population.

(a) The area designated as the plume exposure pathway EPZ for PNPP in the FSAR.
(b) The area which the Staff believes should constitute the plume exposure EPZ for PNPP.
(c) The circular zone surrounding PNPP having a 20-mile radius.
(d) The Mentor Headlands area.
(e) The entire City of Mentor.

Response

Interrogatory 43

In the Staff's opinion, would there ever be a need to order protective actions in any area outside of the plume exposure pathway EPZ proposed by the Applicant in the FSAR? If so, describe the circumstances therein, the areas so affected and the nature of any such protective actions.

Response

Under certain severe accident sequences, it might be necessary to take protective actions beyond the 10 mile plume exposure pathway EPZ. This possibility was discussed in the development of the EPZ distances in NUREG-0396 and in NUREG-0654. It was concluded that the planning undertaken with respect to the plume exposure EPZ would provide a basis for expansion of the emergency response effort. The nature of such protective measures is likely to be sheltering during plume passage followed by relocation of individuals from any contaminated area under the "footprint" of the plume.

Interrogatory 44

Has any consideration been made of the voluntary and spontaneous evacuation of persons within the plume exposure EPZ in the event of an accident at PNPP and how this might affect the ordered evacuation? If so, describe in detail any such study.

Response

FEMA has been requested to respond to this interrogatory.

Interrogatory 45

Has any consideration been made of the possibility of the voluntary and spontaneous evacuation of persons outside of the plume exposure pathway EPZ in the vent of an accident at PNPP and how this might affect the ordered evacuation, especially the support organizations and facilities outside the EPZ? If so, describe in detail any such study.

Response

FEMA has been requested to respond to this interrogatory.

Interrogatory 46

In the Staff's opinion, are there adequate facilities available to shelter simultaneously the total permanent and peak seasonal and transient populations in each of the following areas?

(a) The area designated by the Applicant in the FSAR as the plume exposure pathway EPZ.

(b) The area which the Staff believes should comprise the plume exposure pathway EPZ.

(c) The circular zone surrounding PNPP having a 20-mile radius.

With respect to each of these areas, describe the types of shelter available, indicate the numbers of each type of shelter available and the shielding factors associated with each type, describe the nature and location of the shelter to be used by transient populations, and disclose any assumptions made as to an acceptable level of risk to the public.

Response

The State and local plans have not been completed. In general, however, there is no requirement to have public facilities available to shelter simultaneously the total permanent and peak seasonal and transient populations. If in-place sheltering appreared to be the appropriate response to reduce exposure, transients (e.g., those on beaches) would likely be told to evacuate rather than shelter. FEMA has been requested to respond with respect to the types of in-place shelter in the site area.

Interrogatory 47

Describe in detail any design modifications which would be made to PNPP, Units 1 and 2 to reduce the early and/or delayed fatilities and/or health effects associated with accidents. Specify the type of accident(s), the consequences of which each such modification would reduce, and estimate, for each modification, the extent of reduction for each of the following effects: early fatalities, delayed fatalities, early injuries, delayed injuries, and developmental or genetic birth defects.

Response

Interrogatory 48

In the Staff's opinion, what constitutes an acceptable level of risk to the public surrounding PNPP in the event of an accident? Specifically, what is the uppermost number of each of the following health effects which is acceptable: early fatalities, delayed fatalities, early injuries, delayed injuries, and developmental or genetic birth defects? If your answer varies depending on the type of accident which occurs, provide answers with respect to releases at PNPP equivalent to the BWR-1 to BWR-4 releases defined in WASH-1400.

Response

Interrogatory 49

In the Staff's opinion, what constitutes an appropriate and safe distance from PNPP for the location of reception/mass care centers for evacuees? Describe any other criteria for the location of reception/mass care centers.

Response

The NRC/FEMA guidance on relocation centers in host areas, as stated in Criterion II.J.10.h of NUREG-0654, is that they should be at least 5 miles, and preferably 10-miles, beyond the boundaries of the plume exposure emergency planning zone.

FEMA has been requested to respond to this interrogatory.

Interrogatory 50

In the Staff's opinion, if an accident occurs on a weekday during working hours what percentage of the permanent population within the plume exposure pathway EPZ proposed by the Applicant will be working at locations outside the EPZ, leaving other family members at home without automobiles? Also, what percentage of the automobiles said to be available in the Applicant's Evacuation Study are operable?

Response

The Staff will provide a response at a later date.

Interrogatory 51

Why has the Applicant not submitted separate evacuation time estimates for evacuating special facilities, as required by NUREG-0654, Appendix 4?

Response

Information on special facilities including evacuation time estimates is presented in Appendix D to the Emergency Plan (in Appendix 13A to the FSAR dated May 22, 1981). See Table V-4 of Appendix D under "TransportDependent Population Evacuation Time."

Interrogatory 52

NUREG-0654, Appendix 4 provides that (at p. 4-2), in preparing evacuation time estimates, "The number of permanent residents shall be estimated using the U.S. Census data or other reliable data, adjusted as necessary, for growth." (Emphasis added) In the Staff's opinion, what is the appropriate target date for adjusting population figures for growth: the expected date of initial criticality, or the expected date for the termination of plant operations? Why have unadjusted population data been used to prepare evacuation time estimates? Provide evacuation time estimates using properly adjusted population.

Response

The Staff position is that population estimates projected for the beginning of plant operation should be used for determining evacuation time estimates. Hence, the statement in NUREG-0654, Appendix 4 requesting that latest Census data (which was for 1970 when NUREG-0654 was issued), or other population data, be adjusted for growth. Evacuation time estimates are considered by the Staff to be a dynamic management decision tool and as such are to be periodically updated over the life of the plant to account for changes in population and traffic network. The Applicant used 1980 population estimates in developing the evacuation time estimates presented in the Emergency Plan. The staff has evaluated the population distribution and growth around the PNPP site as reported in Section 2.1 of the SER. This evaluation showed that little growth (from 73, 134 to 74,085) was expected in the resicent al population within 10 miles of the si:c betw.en 1978 and 1983.

Interrogatory 53

The Applicant's FSAR, Appendix 13A, Section 4.2 states that the Ohio DSA has adopted the EPA manual of protective action guidelines, EPA-520/1-75-001, and that recommendations to the State and local government will be based on these PAGs.

(a) Is the Staff aware that this includes the administration of radioprotective drugs, such as potassium iodide?

(b) If so, describe in detail any and all provisions for the purchase, storage, stockpiling, distribution (including public education on proper use of the drug), and effectiveness/side effects monitoring of such durgs.

(c) In the Staff's opinion, would the administration of radioprotective durgs to individuals off-site ever be necessary or desirable in the event of an accident at PNPP? If not, why not? If so, to what radial distance from the site could dissemination of the drugs be necessary? What is the maximum quantity of potassium iodide or other radioprotective drug that could be needed? What repositories in the vicinity of the PNPP site currently stock such drugs and what quantities are maintained?

Response

Protective action guides (PAGs) are projected doses to individuals in the population which act as trigger points to initiate protective actions to minimize the radiation exposure of the general public resulting from an accident. Depending on the type and severity of the accident, several different protective actions or combination of protective actions may be recommended. One such protective action is the administration of potassium iodide to reduce the uptake of inhaled or ingested radioidine by the thyroid gland. The NRC and FEMA staffs have recommended that nuclear power plant licensees as well as State and local governments stockpile radioprotective drugs for thyroid protection in the event of a nuclear accident for emergency workers onsite, emergency workers offsite in the plume exposure EPZ, and institutionalized persons within the plume exposure EPZ whose immediate evacuation may be infeasible or very difficult. The NRC has continued to evaluate the use of potasimum iodide and its administration to the general public within the 10 mile plume exposure EPZ. To date, the NRC has found no compelling reason to recommend the distribution of potassium iodide for this purpose. The plans for use of potassium iodide for offsite emergency workers and institutionalized persons are reviewed by FEMA as this aspect is under the jurisdiction of the State and local governments.

Additional information on this subject has been requested from FEMA.

Interrogatory 54

Explain precisely how each of the following possibilities was accounted for in the preparation of evacuation time estimates for PNPP:

(a) Vehicles breaking down or running out of fuel during the evacuation.

(b) Abandoned vehicles.

(c) Vehicles having insuffieient fuel at the commencement of the evacuation, to the knowledge of their owners.

(d) Disregard of traffic control devices.

(e) Evacuees using inbound traffic lanes for outbound travel.

(f) Blocking of cross-streets at intersections.

Response

A response will be provided at a later date.

Interrogatory 55

In the opinion of the Staff, does the Applicant's FSAR comply with each and every item applicable to BWRs in Regulatory Guide 1.97, Revision 2? If your answer is anything other than unconditionally affirmative, describe in detail every item of noncompliance, the alternative approach proposed by the Applicant, and the safety justification for that alternative approach.

Response

Interrogatory 56

In the opinion of the Staff, what consititues a functional letter of agreement between a utility and off-site emergency response organization? Does the Staff consider the letters of agreement contained in Appendix B of Appendix 13A of the PNPP FSAR to be functional and binding and legal?

Response

A functional letter of agreement is one which briefly describes the response capability of the supporting organization, commits the organization to respond if called upon, and is signed by a proper authority representing the agency. While the format of the sample letters provided in Appendix B to the Emergency Plan is generally acceptable, the Staff has requested the Applicant to update the letters and provide letters of agreement for all offsite agencies relied upon for support. This aspect of applicant's emergency preparedness will be verified prior to startup.

Interrogatory 57

What provisions have been made to ensure the cooperation of the public during a radiation emergency? Specifically, what authority do state and local governments have to force people to evacuate from their homes, to prevent spontaneous evacuation outside the EPZ (and possibly in the area of the reception/mass care centers), to compel the assistance of volunteers in the evacuation, and to control panic and subsequent uncooperative behavior in evacuees?

Response

FEMA will respond to this interrogatory.

Interrogatory 58

In the Staff's opinion, might a nuclear emergency occurring at PNPP ever require the imposition of marital law? If so, what areas around the site might be so affected and for how long?

Response

FEMA will respond to this interrogatory.

Interrogatory 59

Has the State of Ohio Nuclear Power Plant Emergency Response Plan received concurrence from the NRC? If not, specify which portions of the plan do not comply with NRC criteria for concurrence. Provide these criteria.

Response

The NRC formerly concurred in offsite emergency plans, a term which signified an agreement with a voluntary effort on the part of offsite agencies to develop radiological emergency response plans. However, since December 1979, FEMA reviews and evaluates offsite emergency plans and submits its findings and determinations to the NRC. State and local plans, specific for the PNPP site, have not yet been submitted to FEMA. The State of Ohio emergency plan has been reviewed by FEMA in conjunction with the review of emergency preparedness at other nuclear facilities within and adjacent to the State.

Interrogatory 60

NUREG-0654 at p. 13 states that "The range of times between the onset of accident conditions and the start of a major release is of the order of one-half hour to several hours." Section 5.2.4 of Appendix 13A of the PNPP FSAR states that the Emergency Duty Officer, who must be able to respond within 60 minutes, is responsible for recommending protective actions for the general public. In the Staff's opinion, in the event of a major release occurring within ½ hour, does this arrangement provide sufficient time for alerting the public and implementing the appropriate protective actions, including evacuation, such that no members of the general public would receive radiation doses in excess of the limits prescribed by 10 CFR Sec. 20.105?

Response

An individual is required to be onshift at all times who has the authority to immediately and unilaterally initiate any emergency actions including the decision to notify offsite authorities and to recommend protective actions. At PNPP, this individual is initially the Shift Supervisor, who is relieved of these duties as higher levels of plant management assume the functions of Emergency Coordinator according to an established line of succession. The dose limits in 10 CFR 20.105 are applicable during routine operation of the plant. During an emergency situation, protective action guideline doses are used to develop protective action recommendations for the public.

Interrogatory 61

What is the basis of legal authority (regulations, legislation at Federal, State, or local level) governing the cooperation of state, local, or private emergency response organizations with utilities operating nuclear power plants?

Response

The requirement that offsite emergency plans be in place is given in the revised regulations on emergency planning, and is a condition of licensing imposed upon the Applicant. That is, the regulation is not a requirement on offsite governmental authorities but rather is a condition which the Applicant must satisfy before a license is granted. State and local governments have a responsibility to protect the health and safety of their citizens and accordingly develop emergency plans for a spectrum of industrial, transportation, and natural hazards.

Interrogatory 62

Describe in detail any independent monitoring for radiation around the PNPP site. (Independent monitoring here means monitoring by a governmental or private entity that is not an agent of the Applicant.) Include the types of monitors to be used, both mobile and stationary and detection/manufacturer type, manner and frequency of reading/analysis, availability of instantaneous data, type of data link with the responsible agency, name and affiliation of responsible agency, type of meteorological monitors/data input, if any, means of calculating projected doses, and the source of funding of the responsible agency.

Response

The NRC will have in place about 40 thermoluminescent dosimeters (TLDs) prior to plant operation. These are stationary devices which can be retrieved and read to determine integrated exposure at that point.

FEMA will respond with respect to accident monitoring capabilities of offsite authorities.

Statement of Purpose: The following interrogatories relate to the Quality Assurance contention admitted to date. In the event the contention is expanded further interrogatories will be filed. The interrogatories relate to the contention and are designed to provide information and evidence on the hearing.

Interrogatory 63

Provide the name and last known address of the person who made allegations regarding the QC program at PNPP during a telephone call to NRC Region III on May 23, 1978. (See Investigation Report No. 50-440/78-08; 50-441/78-07, p.2) Provide also the name and last known address of the second individual interviewed on May 26, 1978. Provide transcripts of the interviews conducted with these persons on May 24 and 26, 1978.

Response

This interrogatory asks for names and addresses of persons making allegations to Region III, and for the transcripts of the interviews with those persons. As a matter of policy, NRC does not make public the names and addresses of allegers. This is done to protect the individuals from any possibility of retribution for making the allegations. In addition, no transcripts of the interviews were kept, and so, cannot be supplied.

Interrogatory 64

Document each and every instance in which equipment and/or materials not meeting specifications were used "as is". For each case, name all personnel responsible for this decision to "use as is" and their qualifications, list any instance in which an engineering judgement was used in reaching that decision and the basis of that judgement, and indicate any applicable operational experience.

Response

This interrogatory asks for documentation of each and every "use-as-is" disposition at the Perry Plant, including the names and qualifications of all responsible personnel, and also for each instance of engineering judgment, the basis for each judgment, and any "applicable operational experience" (sic). It does not appear that this interrogatory should be addressed to the NRC. At any given site there may be thousands of "use-as-is" dispositions on recorded discrepancies or deficiencies. It would take many man-months on-site to compile the voluminous records which would be required to answer this interrogatory. The NRC does not make the"use-as-is" judgments, but merely reviews dispositioned deficiency reports on a sampling basis to verify that appropriate engineering reviews are being made. The request for "applicable operational experience" does not appear to be meaningful since the Perry units are not yet operational.

Interrogatory 65

Provide all documentation concerning corrective actions taken regarding the improper alignment of the Unit 1 RPV (see Unresolved Item 440/78-12-05). The following questions relate to the closure of above unresolved item in the inspection report and related correspondence dated Nov. 21, 1979.

(a) On what previous experience was the "use as is" decision made by GE safety/reliability personnel based?

(b) Was this decision based on any engineering judgement? If so, provide the basis of that judgement.

(c) Give the names of the GE personnel responsible for that decision.

(d) Provide the qualifications of all persons named in (c).

Response

This interrogatory requests all documentation concerning an identified out-of-alignment condition of the Unit 1 RPV, including the basis of the GE engineering decision, the names and experience of the GE personnel involved in the decision, and the qualifications of the GE personnel. This is another item which should be directed to the licensee for answer. The R-III "unresolved item" on this subject was closed in Report No. 50-440/79-10. The information requested regarding the GE personnel involved in the engineering evaluation is not contained in any R-III records.

Interrogatory 66

Provide all applicable documentation of the interviews with site construction craftsman conducted by Region III personnel, including names of the craftsmen and the NRC interviewers and transcripts, notes, tapes, etc. of the interviews conducted on October 16-18, 1979 and November 14-15, 1979, and at any previous or subsequent time. (See inspection reports and related correspondence: 50-440/79-10 and 50-441/79-10, dated Nov. 21, 1979; 50-440/79-11 and 50-441/79-11, dated Dec. 17, 1979)

Response

A memorandum dated April 2, 1980, to be sent under separate cover, is the only documentation available on the subject interviews. No records of interviewee names were retained. In addition, no transcripts, notes, or tapes of interviews were kept, and so, cannot be supplied. Additionally, IE Inspection Report Nos. 50-440/80-02 and 50-441/80-02 provide further documentation of these interviews. This document is available in the Public Document Room.

Interrogatory 67

What are the flammability requirements for electrical cable and related components, such as connectors, tees, terminations, insulation, etc.? Has the material used at PNPP met these requirements?

Response

The flammability requirements for electrical cabling for Perry are contained in Branch Technical position CMEB 9.5-1, Section C.5.e(3), which requires that, as a minimum, electrical cable construction posses the flame test specified in the current edition of IEEE-383. No flammability requirements exist for components such as electrical tees, connectors, or termination. CE has committed that all electrical cabling which it installs will meet IEEE-383 criteria (see Section 9.5.1.4.5 of the Perry SER, NUREG-0887).

Interrogatory 68

Describe in detail the "fabrication deficiencies" which caused rejection of service water intake structures, as documented in Inspection Report No. 50-440/80-09 and 50-441/80-09, p. 4.

(a) Were any design changes made as a result of this problem?

(b) Provide the names of all personnel involved in this decision and give their qualifications.

(c) Was this decision based on any engineering judgement? If so, give the basis of that judgement.

Response

Licensee reports dated December 7, 1979 and February 12, 1979, to be sent under separate cover, contain all information relevant to these deficiencies available in the Region III files. Region III has no record of the information requested in items (a), (b), and (c). This information should be requested from the licensee.

Interrogatory 69

After it had been discovered that an inspector's initials had been forged on an inspection document (see Investigation Report 50-440/78-08, 50-441/78-07), were any other documents checked for signature authenticity? If not, why not? If so, list all documents so checked, and indicate any which did not meet with approval.

Response

Section III of IE Inspection Report Nos. 50-440/78-15 and 50-441/78-14 identifies the additional documents reviewed for signature authenticity. The licensee reviewed 462 out of 800 embed checklists. In addition, the Region III inspector randomly selected and reviewed the following:

a. Twenty-five Receipt Inspection Nonconformance Reports.

b. Fifteen CQC inspection reports.

c. Seventy-five signatures on Receipt Inspection documents.

d. Fifteen construction quality action requests.

Respectfully submitted,

Michael N. Wilcove
Counsel for NRC Staff

Nathene A. Wright
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Dated at Bethesda, Maryland
this 2nd day of August 1982